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SOME PROCEDURES FOR THE PRODUCTION OF SHORT LIVED RADIOISOTOPES

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SOME PROCEDURES FOR THE PRODUCTION OF SHORT LIVED RADIOISOTOPES

by

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PREFACE

Many radioisotopes are available under readily procurable license and virtually as articles of commerce from contractors to the U. S. Atomic Energy Commission. The cost is nominal and is comparable to other scientific laboratory materials and equipment. It is to be expected that restrictions on traffic in radioactive materials will soon be reduced to those safety precautions necessary for any material potentially injurious to public health and safety.

In the case of short lived radioactive materials, however, it is not feasible to make time-consuming shipment from a nuclear reactor production facility to the place of use. No compromise is possible between high initial radiation intensity with a prohibitive shielding requirement in transport and a usable residual radiation at the time and place of receipt by the user.

It is a fortunate natural phenomena that many of the short lived radioisotopes, which have utility but can never become articles of commerce, can be produced by fluxes of thermal neutrons available from the facilities existing at the U. S. Naval Postgraduate School or potentially available to that institution. It is the purpose of this Report to outline procedures for such production and to discuss some of the economic factors involved.

The guidance and assistance in the preparation of this Report by Professor Edmund A. Milne of the Department of Physics and Professor William W. Hawes of the Department of Metallurgy are gratefully acknowledged.

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CHAPTER I

RADIOACTIVATION WITH THERMAL NEUTRONS

There are many methods of inducing radioactivity from naturally occurring stable isotopes by bombardment with high energy gamma radiation and the several particles: alpha, neutron, proton and deuteron. In most cases these nuclear reactions require radiation or particle energies in one of a few narrow resonance bands. Production of each isotope is a special case when radiation or particles of a particular resonance energy are employed.

This Report will be confined to the use of the much more versatile thermal neutron for which a large number of isotopes have an adequate capture cross section. There are several methods of obtaining fluxes of neutrons of a broad spectrum of energies. The various neutron energies are readily moderated or slowed to thermal energy (0.025 electron volts) by elastic collision with protons of a proton-rich material. Water and paraffin are common materials used for this purpose and are of approximately equal effectiveness. This is an efficient process because each neutron-proton elastic collision reduces the momentum of the neutron according to the difference in its vector velocity and that of the proton. Thus after a few collisions the neutron's momentum and energy loss per collision is greatly reduced and is finally very small when thermal equilibrium with the protons is reached. Hence, neutrons of all initial energies are rapidly attenuated to thermal energy after which energy loss per collision is very small. If it is assumed that the neutrons initially emanate equally in all directions from a point source, they are again

randomly directed from any point after thermalizing; and the flux field thereafter is still subject to approximately spherical divergence. Figure 1 shows the thermal neutron flux distribution from two million electron volt deuterons from a Van de Graaff accelerator employing the $Be^{9}(d,n)\ B^{10}$ reaction [1]. The shape of the curve is independent of initial flux and nearly independent of initial energy distribution. Boyd [2] recommends three centimeters of paraffin for optimum moderation of the entirely different flux and energy spectrum of neutrons from a radium-beryllium source; so it is inferred that flux moderation and moderator thickness attenuation for most neutron sources and proton-rich moderators can be predicted from Figure 1.

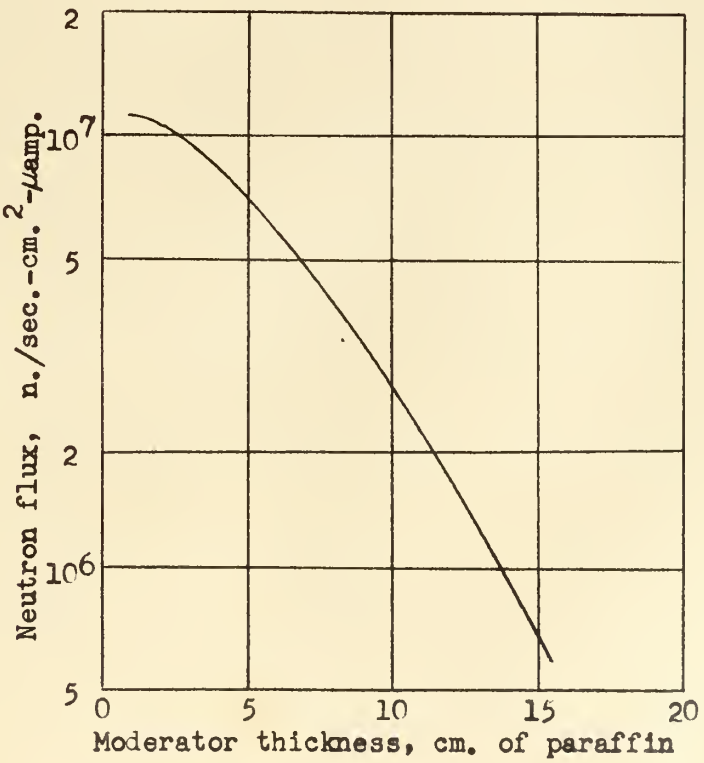


Figure 1. Neutron flux distribution in paraffin

Further details of the production of thermal neutron fluxes will be left to later chapters, and the remainder of this chapter will be devoted to the production of radioisotopes by thermal neutron fluxes.

The basic activation equation for a particular reaction is

$$A_t = (\sigma f N k)(1 - e^{-0.693\ t/T})e^{-0.693\ \Theta/T}$$

where A_t - activity of the radionuclide formed in disintegrations per second at time, Θ seconds, after an irradiation time, t seconds
σ - activation cross section of starting isotope, $cm.^2$
f - thermal neutron flux, neutrons per $cm.^2$ - seconds
N - number of all atoms of the element in the target
k - fractional abundance of the isotope from which the radionuclide is formed
e - the base of natural logarithms, 2.718 ....
t - irradiation time, seconds
Θ - time after cessation of irradiation, seconds
T - half-life of radionuclide, seconds

This relation is valid for target sections thin enough so that the neutron flux is not significantly diminished in traversing the section. Each isotope of each element present in the target must be individually treated and the total activity of the irradiated target is the summation of the activities of all the radionuclides produced. The first term in parentheses is the saturation or maximum activity that can be produced. The second term represents the activity produced less the decay occurring during production. When t is less than 0.15 T, this second term can be abbreviated to 0.693 t/T with less than two percent inaccuracy. The effect of this decay term is shown in Figure 2. The final exponential is to be included when the time of use after cessation of irradiation, Θ, is significant when compared to T. It is the factor of decay after production is stopped.

The product of radiative capture of thermal neutrons by a nucleus of mass M_A is a nucleus of mass M_{A-1} in a highly excited state. With the

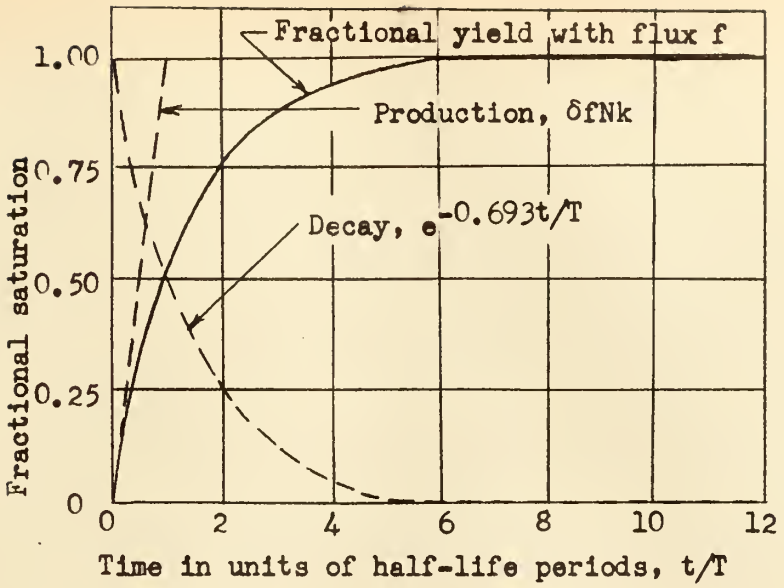


Figure 2. Growth and decay of an isotope during irradiation

exception of a few light elements, e.g., Li, B and N, this great excess of energy is immediately lost by gamma emission [3]. This gamma emission must be considered and shielding provided, if necessary, during the irradiation or production of a radioisotope.

Since isotope production is the topic herein considered, the Table of Isotopes appearing as Appendix I is made up according to the mass and properties of the product isotope while the natural abundance and the thermal neutron capture cross section refer to stable or "raw material" isotope of the target.

CHAPTER II

THE TWO MILLION ELECTRON VOLT VAN DE GRAAFF PARTICLE ACCELERATOR AS A SOURCE OF THERMAL NEUTRON FLUX

The U.S. Naval Postgraduate School has in operation a Type A, Model H, two million electron volt Van de Graaff particle accelerator, manufactured and installed by the High Voltage Engineering Corporation of Cambridge, Massachusetts. The cost of this equipment was $108,000, installed and with accessories, but exclusive of space and utility connection costs. This equipment is considered in this Report to be the primary source of thermal neutron flux at this institution.

In employing the Van de Graaff particle accelerator the best method for producing a flux of thermal neutrons is the Be^{9} (d,n) B^{10} nuclear reaction. The deuteron beam is focussed to about one centimeter diameter and impinged on a three millimeter thickness of beryllium target. A thin foil would provide sufficient beryllium, but the 0.95 inch diameter disk must withstand 15 pounds per square inch pressure, dissipate 20 to 80 watts of heat per hour and have a safety margin for pit-type corrosion from water which backs the disk on one face. The three millimeter thickness was determined after consideration of the properties of beryllium [4].

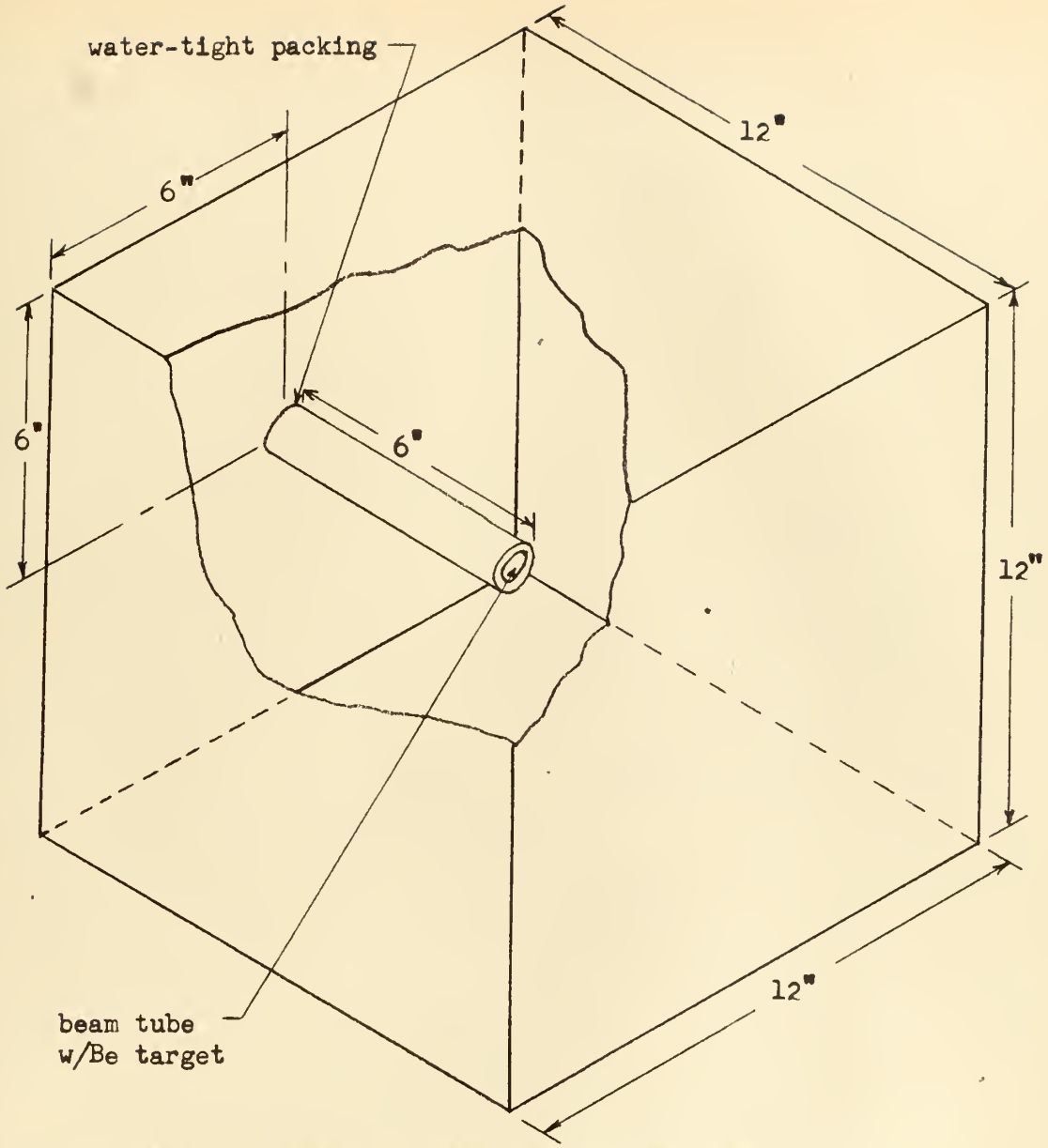
The neutrons emanating from the beryllium are of all energies up to 6.36 Mev, and are more numerous in the forward direction of the deuteron beam by a factor of approximately two. Moderation of these neutrons to thermal energy is by water. Water is selected from two predominant considerations. First, water backing of the beryllium target is desirable for heat dissipation. Second, materials to be irradiated in the thermal neutron flux will be metals unaffected by short exposures to water at room

temperature or will be materials which can be sealed in plastic (but not metal or glass) containers. Furthermore, any sample configuration can be immediately accommodated with a contiguity of moderator. Unless sample configuration is standardized, this would not be the simple case with a solid paraffin moderator.

The thermal neutron flux is directly proportional to the deuteron beam current and varies with moderator thickness traversed as shown in Figure 1. The manufacturers of the Van de Graaff particle accelerator guarantee a beam current of 10 microamperes, and this figure is herein used as a conservative upper limit for the purpose of flux intensity estimation. Reference to Figure 1 indicates that a flux intensity of 10^8 neutrons per second per square centimeter will exist at two to three centimeters from the beryllium target and that this flux will diminish by only a factor of about ten at a distance of 15 centimeters. Figures 3 and 4 show the detailed construction of the assembly to be attached to the view slit collector box of the Van de Graaff. Irradiation samples are to be suspended from the lid of the cadmium box by plastic or other non-metallic supports. It is to be noted that all common laboratory apparatus metals have been avoided since they all have significant cross section to thermal neutron capture, and the radioactivity induced is very long lived in some cases. Cadmium, on the other hand, has a very high capture cross section for thermal neutrons (due to cadmium 113) of 2,350 barns and no resultant radioactivity. One millimeter of cadmium will reduce a thermal neutron flux by a factor of 2×10^{-5} [5].

Appropriate shielding of gamma radiation resulting from the beryllium bombardment must be provided.

The time required to change over the Van de Graaff particle accelerator for deuteron bombardment is a consideration. It is estimated by Professor Milne that thirty-two man hours of supervisory and technician's time are normally necessary. This includes the routine maintenance procedures which are executed whenever the pressure tank is removed.



See Figure 4 for detail of beam tube and Be target mounting

Box made of 2 mm. Cd sheet, lid free and removable, all seams solder sealed and solder removed from inside surfaces.

Fill with distilled water and suspend samples from lid with non-metallic support

Enclose Cd box in reinforcing wooden box and support from floor

Figure 3. Beryllium target holder and cadmium moderator box for attachment to the Van de Graaff particle accelerator

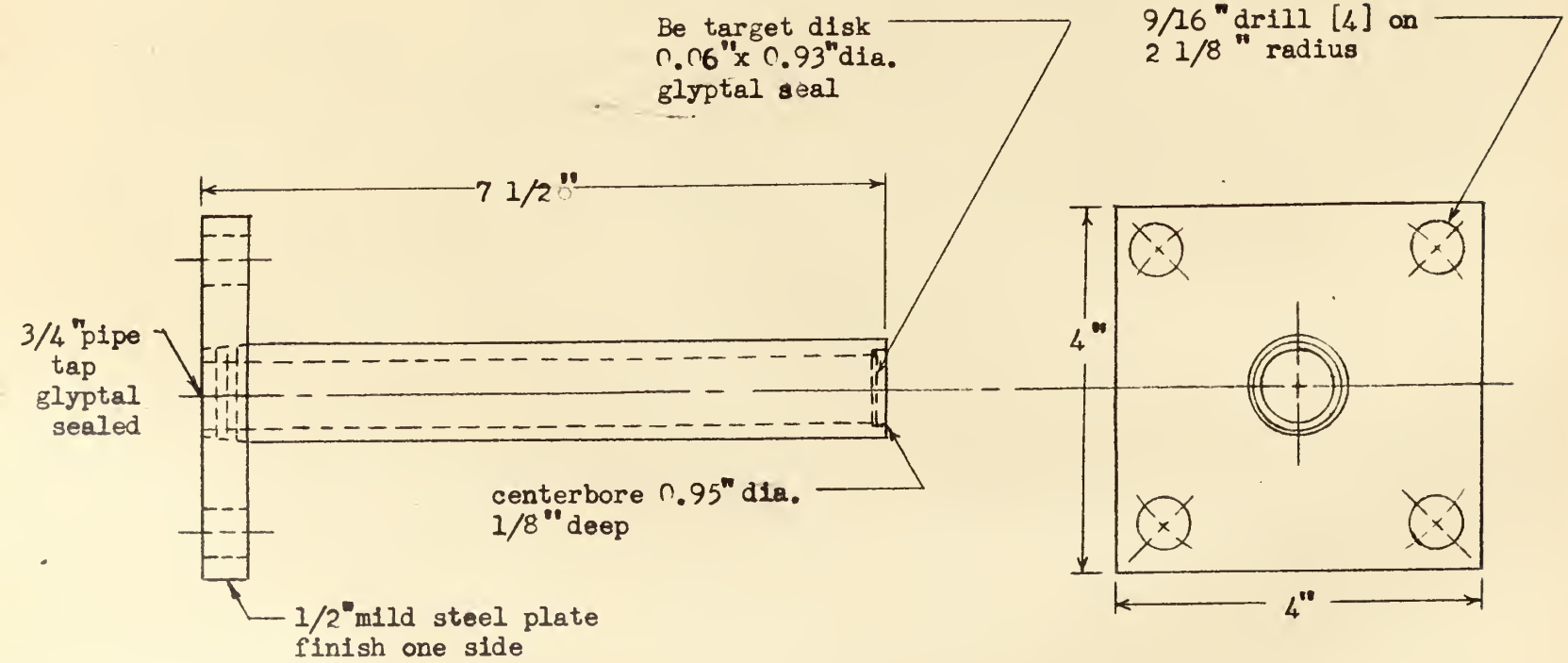


Figure 4. Detail of beryllium target holder

CHAPTER III

LOW LEVEL NEUTRON SOURCES

The five millicurie radium-beryllium neutron source owned by the U.S. Naval Postgraduate School is typical of low level laboratory sources. Five milligrams of radium chloride are intimately mixed with powdered beryllium and encased in a stainless steel cylinder about one-half inch in diameter and one inch long. Radium is a strong alpha emitter and the Be^{9} (α,n) C^{12} nuclear reaction produces a neutron flux which, when moderated to thermal energy, is about 20 neutrons per square centimeter per second. With this flux a parent isotope with a thermal neutron capture cross section of ten barns or more (see Appendix) will yield a detectable amount of radioactive product. There is a requirement for gamma shielding, but the source is otherwise quite convenient. It does not decay appreciably (radium half-life is 1,560 years), and the cost is not excessive. Radium is valued at about $25 per milligram. A source with a larger amount of radium will produce a larger neutron flux in direct proportion to the amount of radium present.

An antimony-beryllium source is available from the Union Carbide Nuclear Company, contract operator for the Atomic Energy Commission [6]. The Be^{9} (γ,n) Be^{8} reaction is utilized here for neutron production. The capsule itself is 1.15 inches in diameter and 1.27 inches long and is priced at $44. Irradiation in a nuclear reactor for 120 days costs $132 and produces 2.6 curies of antimony 124, a very strong gamma emitter with a 60-day half-life. Therein lie the two disadvantages of this source. Shipment must be made in a 700 pound lead container and six inches of lead

shielding are necessary at all times. Semiannual reirradiation is necessary if the activity is to be maintained above 12 percent of its initial value. A thermal neutron flux of 10^5 neutrons per square centimeter per second is to be expected from this source when newly irradiated.

A polonium-beryllium neutron source is also available at a cost of about $1,000 for a one curie strength. A thermal neutron flux of 3×10^4 neutrons per square centimeter per second is available from this source. The Be^9 (α,n) C^{12} reaction is employed, and there is insignificant gamma emission, which is a great advantage. Another advantage over the Sb-Be source is the longer half-life of 140 days.

In utilizing any of the above low level sources, the cadmium box of Figure 3 may be employed for water moderation and neutron shielding. The side hole for the beam tube is blanked off with a cadmium cover, and the source is supported on a plastic pedestal at the center of the box. The samples to be irradiated are suspended from the lid of the box as before. In the case of the Sb-Be source there is still the requirement for heavy lead shielding or remote handling facilities.

CHAPTER IV

THERMAL NEUTRON ACTIVATION ANALYSIS

This closely allied subject merits mention in this Report since a large part of the material used herein [1,2,7,8,9] had activation analysis as its primary investigative objective. W.W. Meinke [10] compares sensitivity of neutron activation analysis with other methods of trace-element detection. His comparison shows that even the intermediate level of thermal neutron flux available from the two million electron volt Van de Graaff gives superlative results in the favorable cases of a few elements with very high thermal neutron capture cross section.

The Atomic Energy Commission contract operator offers neutron activation analysis service [6]. This analysis method is used at the Oak Ridge laboratories in the selection of materials for isotope production and as a control for chemical separations.

Although no publication of results has been made for obvious commercial competitive reasons, it has been announced that petroleum company laboratories are investigating the method of neutron activation analysis for oil well logging. High Voltage Engineering Corporation has developed a 200 kilovolt Van de Graaff accelerator only three inches in diameter and 47 inches long for use in the standard 3-5/8" well logging instrument assembly casing. The deuteron-tritium reaction is used for neutron production[11].

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APPENDIX

TABLE OF ISOTOPES WHICH CAN BE PREPARED BY THERMAL NEUTRON FLUX OF 10^8 NEUTRONS PER SQUARE CENTIMETER PER SECOND

This Table is an extract from a table prepared by Burrill and Gale [1] who used data issued by the National Bureau of Standards. 12 .

The Element column shows the isotope produced. In many cases certain isotopes of a naturally occurring elemental target will produce only small activity contributions to the product, but these may not be negligible in total radiation of all types from the product. For example: A pure beta emitter of short half-life is desired. In early stages of product decay this might be the completely predominate situation, but a small amount of long lived activity consisting of other beta energies and some gamma radiation could obscure the later measurement of the decayed product. Hence, every isotope of an element is listed which yields a detectable amount of activity after ten hours of irradiation. A ten percent counting efficiency is assumed.

The Activation Cross Section and Fractional Abundance are for the parent isotope in the naturally occurring element. Reference to more recent work as it appears in publication may refine the data on cross sections.

The Modes of Decay and associated energies are presented clear of parentheses if the probability of their correctness is 50-100 percent, in parentheses if the correctness probability is 15-50 percent and not tabulated if the correctness probability is less than 15 percent in either decay scheme or energy values. Again these decay data will be the subject of refinement by later investigations.

The Specific Activity after one hour of irradiation is presented as a comparison of the activities producible. The comparison is invalid in cases where one hour represents less than three half lives of any of the isotopes under consideration since longer irradiation times are quite feasible.

An approximate "factor of merit" for comparison for possible production can be obtained as the quotient of the cross section divided by the half-life.

Meinke and Anderson [8] plot activation cross section versus half-life, however the cross section data they used is (according to more recent data) grossly in error for many cross sections of less than 0.1 barn. On such a plot an area can be determined containing the point plots of all isotopes producible with a particular thermal neutron flux and irradiation time limitation.

Table of Isotopes Which Can Be Prepared by a Thermal Neutron Flux of 10^8 n./ $cm.^2$ - sec.

Element			Activation Cross Section barns	Fractional Abundance	Half-life		Mode of Decay Mev		Specific Activity after 1 hr
	Z	A	σ	k	T		beta β	gamma γ	Act
A	18	41	1.2	.9961	1.83	h	1.24	1.3	5.71×10^5
Ag	47	108	30.	.5135	2.3	m	2.8	-	8.7×10^6
		110	96.	.4865	24.2	s	2.8	-	2.6×10^7
		110	2.3	.4865	270	d	0.09 (0.59)	0.9,0.66(1.4)	6.5×10
Al	13	28	0.18	1.00	2.4	m	3.01	1.80	4.0×10^5
As	33	76	4.3	1.00	26.8	h	3.04(2.49)(1.29)	(1.2)(0.57)	9.0×10^4
Au	79	198	96.	1.00	2.69	d	0.97	0.41	3.1×10^5
Ba	56	131	6.	.0010	11.7	d		0.49(0.37)	6.8
		139	0.51	.7166	85.	m	2.27(e 0.12)	(0.16)	6.2×10^4
Bi	83	210	0.015	1.00	5.0	d	1.17	-	2.5×10
Br	35	80	8.9	.5052	4.4	h	-	0.49 0.037	6.0×10^5
		80	3.0	.5052	18.	m	2.0	0.5	1.0×10^6
		82	2.3	.4948	35	h	0.47	1.35 0.79 0.55	1.7×10^4
Ca	20	41	0.0002	.9692	10	d			.83
		45	0.63	.206	152	d	0.25	-	3.4
Cd	48	107	1.0	.0121	6.7	h		0.09 Ag - Kx	6.9×10^2
		111	0.2	.1239	48.6	m	-	0.247 0.147	8.0×10^3
		115	0.14	.2886	43.	d	1.7	0.5	1.4×10
		115	1.1	.2886	2.83	d	0.46(1.1)	0.52	2.0×10^3
		117	1.4	.0758	2.83	h	1.7 1.3	-	1.2×10^4

	Z	A	σ	k	T	β	γ	Act
Ce	58	141	0.31	.8848	28 d	0.41 (0.56)	0.14	1.2×10^2
		143	0.95	.1107	33 h	1.35	0.5	9.3×10^2
Cl	17	36	53	.754	2×10^6 y	0.7	-	2.4×10^3
		38	0.6	.246	37.5m	4.81(2.77)(1.11)	(1.6)(2.15)	1.6×10^5
Co	27	60	22.5	1.00	5.3y	0.31	1.17 1.33	3.5×10^2
		60	0.73	1.00	10.7 m	(1.56)	0.05	7.2×10^5
Cr	24	51	16.2	.0431	26.5 d	K	V-Kx	9.0×10^3
		55	0.006	.0238	1.3 h	1.4	-	6.7×10
Cs	55	134	0.016	1.00	3.15h	2.4	0.128	1.4×10^3
		134	25.6	1.00	2.3 y	0.65(0.09)	0.79 0.6(0.56)	4.0×10^2
Cu	29	64	4.3	.691	12.88h	-0.57 +0.65 K	1.35 Ni-Kx	1.5×10^5
		66	2.1	.309	4.34m	2.58	1.32	6.0×10^5
Dy	66	165	2700	.281	1.25m	-	0.10	2.71×10^8
		165	2700	.281	2.42h	1.25 0.88 0.42	0.91 0.76 0.36	6.8×10^8
Er	68	171	7.	.142	7.5 h	1.05(0.67)	0.3 0.113(0.81)	3.3×10^4
Eu	63	152	1530	.4777	9.2 h	1.8 K	0.72	2.2×10^7
		152	2500	.4777	5.3 y	0.9 (1.7) K	1.2 0.3	7.2×10^4
		154	240	.5223	5.4 y	0.3 (0.7)	1.2	1.8×10^2
F	9	20	0.01	1.00	12. s	5.1	2.2	3.2×10^4
Fe	26	55	2.1	.0584	2.9 y	K	Mn - Kx	3.0
		59	0.32	.0031	46. d	0.46 0.26	1.3 1.1	.61
Ga	31	70	1.5	.602	20.5 m	1.65	-	6.9×10^5
		72	3.4	.398	14.25h	(1.00)(0.74)(0.56)	0.84 0.63(2.18)	5.4×10^4

	Z	A	σ	k	T		β	γ	Act
Gd	64	153	125	.0021	200	d	K	0.10 Eu - Kx	1.4×10
		159	4.5	.2475	17.9	h	0.85	0.055 0.35	1.6×10^4
		161	0.17	.2177	3.63	m	1.5	0.37	1.4×10^4
Ge	32	71	0.45	.2055	11.4	d	K	Ga - Kx	2.0×10^2
		75	0.40	.3674	89	m	1.2	-	4.4×10^4
		77	0.085	.0767	12	h	1.7	0.5	3.0×10^2
		77	0.095	.0767	59	s	2.8	-	5.8×10^3
Hf	72	181	10.	.3511	46	d	0.46	0.48 (0.34)	7.2×10^2
Hg	80	197	3100	.0015	64	h	K	0.077 Au - Kx	1.5×10^4
		199	60	.010	43	m	-	0.37 0.15	1.3×10^6
		203	2.4	.296	48	d	0.21	0.29	1.2×10^2
		205	0.34	.067	5.5	m	1.62	-	6.7×10^3
Ho	67	166	67	1.0	27.2	h	1.8	0.07	6.1×10^5
I	53	128	6.8	1.0	24.98	m	2.02	(0.43)	2.6×10^6
In	49	114	61	.0423	48	d	-	0.19	8.3×10^2
		114	2	.0423	72	s	1.98	-	4.5×10^4
		116	145	.9577	53.93	m	0.85	1.31	3.8×10^7
		116	52	.9577	13	s	2.95	-	2.6×10^7
Ir	77	192	260	.385	1.5	m	-	0.06	3.2×10^7
		192	1000	.385	70	d	0.67	0.21 0.14	5.0×10^4
		194	130	.615	19.0	h	2.18 0.48	1.43 0.38	9.0×10^5
K	19	42	1.0	.067	12.44	h	3.57(2.05)	(1.51)	5.4×10^3
Kr	36	79	0.27	.0034	34	h	K	0.2 Br - Kx	1.5×10
		85	0.096	.5699	4.5	h	1.0	0.17	5.1×10^3
		85	0.06	.5699	9.4	y	0.74	-	.2
		87	0.064	.1743	74	m	3.2	-	3.3×10^3

	Z	A	σ	k	T		β	γ	Act
La	57	140	9.0	.9991	40.0	h	1.32 (1.67)	1.61 (0.85)	6.8×10^4
Lu	71	176	30.	.975	3.7	h	1.04	-	1.7×10^6
		177	3200	.025	6.6	d	0.47(0.37)(0.17)	(0.32)(0.21)(0.11)	1.2×10^5
Mg	12	27	0.048	.1129	10.2	m	1.8 (0.9)	0.84 (1.01)	1.3×10^5
Mn	25	56	12.8	1.0	2.59	h	2.86(1.05)(0.73)	0.84(2.13)(1.81)	3.1×10^6
Mo	42	93	0.2	.1586	6.7	h	K	1.7 0.7 0.3Nb-Kx	2.2×10^3
		99	0.3	.2375	67	h	1.05 0.21	0.75 0.14	4.7×10^2
		101	0.5	.0962	14.6	m	2.2 1.0	0.9 0.3	2.8×10^4
Na	11	24	0.5	1.0	14.8	h	1.39	2.76 1.38	6.1×10^4
Nb	41	94	1.2	1.0	6.6	m	-	0.04	7.8×10^5
Nd	60	147	1.5	.1718	11.0	d	0.90 (0.4)	(0.58)	2.4×10^3
		149	2.4	.0572	1.7	h	1.5	-	1.8×10^4
Ni	28	59	4.2	.674	5×10^4	y	K	Co - Kx	4.7×10^{-3}
		63	15.	.038	300	y	0.05	-	.14
		65	2.6	.008	2.50	h	2.10(1.01)(0.6)	(1.5)(1.1)	5.3×10^3
Os	76	191	8	.264	15	d	0.142	0.04	1.0×10^3
		193	1.6	.410	32	h	1.15 e^- 0.14	1.58	6.0×10^3
P	15	32	0.2	1.0	14.3	d	1.71	-	7.9×10^2
Pb	82	209	0.00045	.523	3.32	h	0.68	-	1.3 x 10
Pd	46	109	12.1	.268	13.2	h	0.95	-	9.3×10^4
		111	0.39	.135	26	m	3.5	-	2.3×10^4

Z	A	σ	k	T		β	γ	Act
Pr 59	142	11	1.00	19.3	h	2.14 0.21	1.9 0.13	1.7×10^5
Pt 78	193	150	.008	3.3	d	e^- 0.11	1.5 0.13	30
	197	1.1	.254	18	h	0.65	-	3.4×10^3
	199	3.9	.072	31	m	1.8	-	6.2×10^4
Rb 37	86	0.72	.728	19.5	d	1.8 (0.71)	(1.08)	5.4×10^2
	88	0.12	.272	17.8	m	5.0	-	2.1×10^4
Re 75	186	101	.3707	3.8	d	1.07	-	9.0×10^4
	188	75	.6293	18	h	2.1	1.4 0.48 0.15	5.7×10^5
Rh 45	104	11.8	1.00	4.4	m	e^- 0.069	0.8 Ru - Kx	6.9×10^6
	104	137	1.00	42	s	2.6	0.95 0.18 0.04	8.0×10^7
Ru 44	97	0.01	.06	2.8	d	K	0.23 Tc - Kx	4.0
	103	1.2	.3134	41	d	0.66 0.35	0.55 0.3	1.5×10^2
	105	0.67	.1827	4.5	h	1.3	0.75	9.0×10^3
S 16	35	0.26	.0415	87.1	d	0.16	-	6.8
	37	0.14	.0002	5.04	m	1.6 (4.3)	(2.6)	37
Sb 51	122	6.8	.5725	2.8	d	1.36 1.94	0.57	2.0×10^4
	124	0.03	.4275	21	m	-	0.02	5.1×10^3
	124	2.5	.4275	60	d	0.68 (0.5)	1.7 0.60 (0.7)	2.4×10^2
	124	0.03	.4275	1.3	m	3.2	0.012	6.3×10^3
Sc 21	46	10	1.00	20	s	-	0.18	1.4×10^7
	46	22	1.00	85	d	0.36	1.12 0.89	1.0×10^4

Z		A	σ	k	T		β	γ	Act
Se	34	75	24	.0087	127	d	K	0.12 0.10 As-Kx	39
		81	0.03	.4982	58	m	-	0.01	5.5×10^3
		81	0.5	.4982	18	m	1.5	-	1.7×10^5
		83	0.047	.0919	67	m	3.4	-	3.2×10^3
		83	0.004	.0919	25	m	1.5	1.1 0.37 0.17	2.2×10^2
Si	14	31	0.12	.0305	2.8	h	1.5	-	1.6×10^3
Sm	62	153	280	.2663	47	h	0.78	0.10 0.07	4.3×10^5
		155	6	.2253	25	m	1.9	0.3	4.3×10^5
Sn	50	113	1.1	.0090	105	d	K	0.8 In-Kx	.15
		123	0.3	.0478	47	m	1.26K	0.153 In-Kx	4.4×10^3
		125	0.57	.0611	9.5	m	2.04 1.17	1.86 0.32	1.7×10^4
		125	0.15	.0611	10	d	2.38	-	13
Sr	38	87	1.3	.0986	2.7	h	K	0.37 Rb-Kx	2.1×10^4
		89	0.005	.8256	55	d	1.50	-	1.5
Ta	73	182	20.6	1.00	113	d	0.525	1.24 1.22 1.13	1.7×10^3
		182	0.034	1.00	16.2	m	0.2	0.18	1.0×10^4
Tb	65	160	22	1.00	71	d	0.88 0.55	0.086	3.4×10^3
Te	52	125	5	.0461	58	d	-	0.109 0.035	54
		127	0.073	.1871	90	d	-	0.08	2.1
		127	0.78	.1871	9.3	d	0.80	-	5.4×10^3
		129	0.015	.3179	32	d	-	0.11	2.0
		129	0.13	.3179	72	m	1.8	0.3	9.0×10^3
		131	0.008	.3449	30	h	-	0.177	30
		131	0.22	.3449	25	m	1.8	-	2.8×10^4

	Z	A	σ	k	T		β	γ	Act
Ti	22	51	0.14	.0534	6	m	1.6	-	9 x 10^3
Tl	81	204	7.5	.292	2.7	y	0.78	-	21
		206	0.11	.7054	4.23	m	1.65	-	2.3 x 10^4
Tm	69	170	118	1.00	127	d	0.97 (0.88)	(0.08)	9.6 x 10^3
V	23	52	5	1.00	3.74	m	2.7	1.45	3.9 x 10^6
W	74	185	2.1	.306	74	d	0.428	0.134	83
		187	37.2	.284	24.1	h	0.62 (1.31)	0.48 0.13	9.7 x 10^4
Xe	54	133	0.2	.2694	5.27	d	0.315	0.08	1.4 x 10^2
		135	0.2	.1053	12	m	-	0.52	9.7 x 10^3
		137	0.15	.0894	3.9	m	4.0	-	5.9 x 10^3
Y	39	90	1.2	1.00	62	h	2.18	-	9.0 x 10^3
Yb	70	169	1.8 x 10^4	.0014	33	d	K	0.3 0.2 Tm-Kx	7.9 x 10^3
		175	50	.318	4.1	d	0.5 0.13	0.35	4.0 x 10^4
		177	5	.127	2.0	h	1.3	0.15	6.4 x 10^4
Zn	30	65	0.5	.4887	250	d	K	1.11 Cu-Kx	28
		69	0.9	.1856	13.8	h	-	0.44	7.5 x 10^3
		69	0.09	.1856	57	m	0.86	-	7.8 x 10^3
		71	0.09	.0062	2.2	m	2.1	-	5 x 10^2
Zr	40	93	0.27	.1711	40	y			.06
		95	0.1	.1740	65	d	0.39	0.73	4.8
		97	0.29	.0280	17	h	2.2	0.8	2.0 x 10^2

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